NON-PUBLIC?: N
ACCESSION #: 9112130056
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Byron, Unit 2 PAGE: 1 OF 04

DOCKET NUMBER: 05000455

TITLE: Reactor Trip on Low-2 Steam Generator Level during Startup due to
D-5 Steam Generator Level Control
EVENT DATE: 11/07/91 LER #: 91-005-00 REPORT DATE: 12-09-91

OTHER FACILITIES INVOLVED: None DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 010

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: W. Scheffler, Technical Staff, TELEPHONE: (815) 234-5441
Ext. 2378
R. Wegner, Operating Engineer, Ext. 2206

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

At 0357, on 11/07/91, after Unit 2 startup and during preparation for the turbine roll, a Low-2 Steam Generator (S/G) level on the 2A S/G was received which initiated a Reactor Trip. All safeguard actuation features functioned as designed.

The Low-2 level in the Steam Generator was the result of difficulty in controlling the Westinghouse D-5 S/G level at low power and due to leak by of the main Feedwater Regulating Valve for the 2A Steam Generator. The level instrument taps in the Steam Generator are being moved during the next refueling outage to reduce this level instability. The Main Feedwater Regulatory valve was adjusted before the next startup. A revision will be made to the startup procedure to verify position of this valve for each Steam Generator.

This event is reportable in accordance with 10CFR50.73 (a)(2)(iv), any event or condition that results in a manual or automatic actuation of any Engineered Safety Feature.

(0836R/VS-2)

END OF ABSTRACT

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 11/07/91 / 0357

Unit 2 MODE 1 - Power Operation Rx Power 10%
RCS AB! Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On November 7, 1991 at 0357, Byron Unit Two experienced a Reactor Trip from a Low-2 Steam Generator (S/G) level signal on the 2A S/G, including a Feedwater Isolation, S/G Blowdown Isolation, and Auxiliary Feedwater Actuation. A Nuclear Station Operator (NSO) (RO, licensed) had established manual feedwater tempering flow to the Steam Generators, and was preparing to roll the Main Turbine per 2BGP 100-3, "Power Ascension 5% to 100%". After opening the 2FW039 valves at 0339, an unexpected increase in S/G water levels to 75% Narrow Range was noted. The 2FW039 valves were then closed and an attempt was made to stabilize the S/G levels by initially decreasing and then increasing tempering flow. The 2A S/G level continued to decrease until the Low-2 level Reactor Trip setpoint was reached. The Reactor automatically tripped at 0357. The NRC (Bethesda) was notified of the Reactor trip (event number 22183).

This event is reportable in accordance with 10CFR50.73 (a)(2)(iv), any event or condition that results in manual or automatic actuation of any Engineered Safety Feature.

C. CAUSE OF EVENT:

One cause of the Low-2 level in the 2A Steam Generator was the difficulty in controlling level in the Unit 2 (Westinghouse D-5) Steam Generators at low power levels. The difficulty in controlling level is caused by the small span of the narrow range water level instrumentation on the Unit 2 Steam Generators. This narrow range on Unit 2 is approximately 60% of the more stable Unit 1 (Westinghouse D-4) Steam Generator range. Level deviations in the Unit 2 S/Gs seem amplified when compared to Unit 1 S/G's as a result of the smaller narrow range water level span.

The D-5 S/G level tap locations are located above the transition cone in order to eliminate a 10% error found by Westinghouse during its design testing. This error would have occurred if the Model D-5 had been given the same tap locations as the Model D-4 (U-1). In the Model D-4 the narrow range lower taps are located below the transition cone. In the Model D-4 the velocity head (from the downflow of the recirculation water) below the transition cone is far less than the D-5. For this reason, the D-4 has a 2% error in level indications and the error is taken into account in the D-4 trip setpoints. These different tap locations on the two model Steam Generators, have made the Model D-5 Steam Generators more difficult to control.

The D-5 design increases the effect of the shrink/swell phenomenon during low power operation. This phenomenon creates an initial delay or even a response opposite to the anticipated long term effect. Thus, if an Operator or control system senses the water level dropping below the setpoint and reacts by increasing feedwater flow, the immediate effect will tend to decrease the water level. In other words, the wide range level indication shows what the change will be, while the narrow range shows the immediate effect of the change.

(0836R/VS-3)

C. CAUSE OF EVENT: (continued)

Another cause was leakby on the 2FW510 (main feedwater regulating valve), which caused an unexpected addition of feedwater to the 2A Steam Generator when the 2FW039 valves were opened. The 2FW510 was later found by the Instrument Maintenance Department to be approximately 5% off of its seat.

D. SAFETY ANALYSIS:

All safeguard equipment functioned as designed which resulted in placing the Reactor in a safe condition. The 2A S/G Low-2 Level caused a Reactor Trip, Feedwater Isolation, S/G Blowdown Isolation, and an Auxiliary Feedwater Start. The safety significance would be the same if the same events occurred under different initial conditions.

E. CORRECTIVE ACTIONS:

The Unit 2 (D-5) Steam Generator low power water level instability will be corrected during the next refueling outage (B2R03), by relocating the narrow range level lower taps to a position below the transition cone. Modification M6-2-89-033 will allow the Model D-5 Steam Generators to respond more like the Model D-4 Steam Generators, which will decrease the possibility of future Reactor trips due to level instability. The modification will lower the lower range sensing lines and result in the following changes:

1) Lower tap elevation changes from 438'2.375" to 429'5.25" (lowered by approximately 9 feet).

2) The current span will increase from 128 inches to 233 inches (Unit 1 span is 233 inches).

3) The Steam Generator level setpoints will be as follows:

Current Proposed
Unit 1 (%/in) Unit Two (%/in) Unit Two (%/in)

High-High 81.4/523 78.1/538 80.8/522

Nominal 63.0/480 50.0/502 63.7/482

Low-Low 40.8/428 17.0/460 36.3/418

To correct the leakby of the Main Feedwater Regulating Valve, Nuclear Work Requests B89087, 89088, 89090, and 89089 were written on the four individual regulating valves to check for leakby. The valves were adjusted/checked by the Instrument Maintenance Department, before the subsequent startup. A step will be added to 1/2BGP 100-2, "Plant Startup", to locally check the position of the Main Feedwater Regulating Valves prior to opening the 1/2FW039 valves. NTS Item # 455-200-91-02300-01 tracks this action.

(0836R/VS-4)

F. PREVIOUS OCCURRENCES:

LER NUMBER TITLE

87-002 Reactor Trips and Feedwater Isolations due to
Operator Difficulty in Controlling Steam
Generator Level Transients at Low Power.

88-007 Feedwater Isolation Actuations due to Steam
Generator Preheater Bypass Valve Failure to
Open.

90-003 P-14 Feedwater Isolation due to Inability to
Control Level in D-5 Steam Generators at Low
Power Levels.

G. COMPONENT FAILURE DATA:

Not Applicable.

(0836R/VS-5)

ATTACHMENT 1 TO 9112130056 PAGE 1 OF 1

Commonwealth Edison
Byron Nuclear Station
4450 North German Church Road
Byron, Illinois 61010

December 6, 1991

Ltr: BYRON 91-0975

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Dear Sir:

The enclosed Licensee Event Report from Byron Generating Station is being
transmitted to you in accordance with the requirements of
10CFR50.73(a)(2)(iv).

This report is number 91-005; Docket No. 50-455.

Sincerely,

R. Pleniewicz

Station Manager
Byron Nuclear Power Station

RP/DK/mw

Enclosure: Licensee Event Report No. 91-005

cc: A. Bert Davis, NRC Region III Administrator
W. Kropp, NRC Senior Resident Inspector
INPO Record Center
CECo Distribution List

(0836R/VS)

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